WASTE CHARACTERIZATION, CLASSIFICATION, AND SHIPPING SUPPORT TECHNICAL BASIS DOCUMENT

for

Battelle Columbus Laboratories Decommissioning Project (BCLDP) West Jefferson North Facility

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Prepared By:

C. W. Skapik
Research Scientist

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Date

This document, DD-98-04, *Waste Characterization, Classification, and Shipping Support Technical Basis Document*, has been reviewed and concurred with and approved by the following:

Concurred With:

S. J. Maheras
Senior Research Scientist

11/18/2002
Date

Concurred With:

S. R. Mawalkar
Senior Research Scientist

11/15/2002
Date

Approved By:

S. D. Schmucker
Project Manager, Low Level Waste (LLW)

11/18/02
Date

J. H. Eide
Manager, Transuranic (TRU) Waste Program

11/18/02
Date

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ACRONYMS

AK	Acceptable knowledge
BCLDP	Battelle Columbus Laboratories Decommissioning Project
CH-TRU	Contact-handled TRU
CoC	Certificate of Compliance
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
JN	Battelle's West Jefferson North Site
LHS	Latin hypercube sampling
LLW	Low-level waste
ORIGEN2.1	Oak Ridge Isotope Generation and Depletion Code
QA	Quality assurance
QAD-CGGP-A	Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor (CGGP – Combinatorial Geometry with Geometric Progression)
RH-TRU	Remote-handled TRU
SNF	Spent nuclear fuel
SNM	Special Nuclear Material
SWAC	Hanford Site Solid Waste Acceptance Criteria
TRU	Transuranic waste
V&V	Verification and validation
WCP	Waste Certification Program
WIPP	Waste Isolation Pilot Plant

1.0 INTRODUCTION

1.1 OVERVIEW

The purpose of this document is to establish the technical basis for identifying and quantifying the isotopes in the radioactive remote-handled transuranic (RH-TRU) waste and low level waste (LLW) generated by the Battelle Columbus Laboratories Decommissioning Project (BCLDP) at Battelle's West Jefferson North (JN) site. It is divided into the following sections:

1. Definition of the JN Standard Isotopic Mix—Technology limitations and the need to limit personnel exposure associated with the high exposure rate of the waste inventory stored at JN-1 limit the feasibility of direct sampling for the purpose of radiologically characterizing this waste. Since direct sampling is not feasible for all of the waste shipped from the West Jefferson North Site, a "JN Standard Isotopic Mix" is established in this section based upon acceptable knowledge (AK), sampling, and computer modeling a representative source term. Gamma exposure rates from waste packages are compared with the "standard," and isotopic inventories in the waste packages are calculated from that comparison.

2. Method for Estimating Waste Container Radioisotopic Inventories—This section sets forth the method for identifying and quantifying the isotopic inventory for waste based upon the external gamma exposure rate. For RH-TRU, the waste addressed by this section is packaged into 55-gallon drums with liners. The liners are filled remotely inside the hot cells in the JN-1 building. Once filled, each liner will receive two (2) designations as described in this section. The first designation will be either contact-handled (CH) or remote-handled (RH) waste. The second designation will be either TRU waste or LLW. Waste designated as LLW is segregated by the BCLDP from the TRU waste and is handled under the BCLDP LLW program.

The general process for identifying waste container contents—for both RH-TRU and LLW—is as follows:

Step 1: The external gamma exposure rate for each container is measured.

Step 2: The measured rate is compared with that of a known activity modeled using the radiation shielding computer program Microshield for RH-TRU and QAD for LLW.

Step 3: By setting the ratios of exposure rate to activity equal for the unknown drum (shown in Equation 1 below) and the modeled drum, the total activity within the waste container is calculated.

$$\frac{A_{DRUM}}{\dot{X}_{DRUM}} = \frac{A_{MS/QAD}}{\dot{X}_{MS/QAD}} \qquad \text{(Equation 1)}$$

Step 4: The total activity within the waste container is then split up into constituent isotopes by multiplying the total activity by the JN standard composite in Table 1
.

Table 1: The JN Standard Isotopic Mix

Radionuclide	Basis	Activity Ratio	Normalized Activity (Ci/1 Ci)	Ratioed Standard Deviation	Normalized Standard Deviation
Ac-227*	O	1.106E-10	3.730E-11	2.578E-12	8.693E-13
Am-241	S	5.013E-02	1.690E-02	5.931E-03	2.000E-03
Am-242m	O	1.001E-04	3.375E-05	1.137E-06	3.834E-07
Am-243	O	3.746E-04	1.263E-04	4.216E-06	1.422E-06
Ba-133	O	6.222E-30	2.098E-30	5.368E-31	1.810E-31
Be-10*	O	7.465E-11	2.517E-11	3.803E-13	1.282E-13
C-14	O	1.871E-05	6.309E-06	1.741E-07	5.871E-08
Cd-113m	O	4.065E-04	1.371E-04	3.339E-06	1.126E-06
Cf-249*	O	6.009E-10	2.026E-10	2.629E-11	8.865E-12
Cf-250	O	1.410E-09	4.755E-10	6.127E-11	2.066E-11
Cf-251*	O	2.381E-11	8.029E-12	1.149E-12	3.875E-13
Cl-36*	O	1.870E-07	6.306E-08	1.733E-09	5.844E-10
Cm-243	O	2.628E-04	8.862E-05	3.629E-06	1.224E-06
Cm-244	S	4.036E-02	1.361E-02	4.944E-03	1.667E-03
Cm-245	O	6.419E-06	2.165E-06	5.384E-08	1.816E-08
Cm-246	O	2.199E-06	7.415E-07	3.705E-08	1.249E-08
Cm-247	O	1.023E-11	3.450E-12	2.744E-13	9.253E-14
Cm-248	O	4.206E-11	1.418E-11	1.606E-12	5.416E-13
Cm-250*	O	9.677E-18	3.263E-18	5.028E-19	1.696E-19
Co-60	S	3.096E-01	1.044E-01	2.784E-01	9.388E-02
Cs-134	S	4.113E-03	1.387E-03	3.073E-04	1.036E-04
Cs-135	O	4.986E-06	1.681E-06	3.467E-08	1.169E-08
Cs-137		1	3.372E-01	0	0
Eu-150*	O	1.687E-10	5.689E-11	8.858E-13	2.987E-13
Eu-152	O	3.752E-05	1.265E-05	7.151E-07	2.411E-07
Eu-154	S	1.543E-02	5.203E-03	9.435E-04	3.182E-04
Gd-152*	O	5.894E-18	1.988E-18	6.035E-20	2.035E-20
I-129	O	4.757E-07	1.604E-07	2.195E-09	7.402E-10
K-40	O	7.404E-14	2.497E-14	6.586E-16	2.221E-16
Mo-93*	O	1.049E-08	3.537E-09	9.448E-11	3.186E-11
Nb-94	O	2.265E-09	7.638E-10	2.075E-11	6.997E-12
Ni-59	O	7.336E-06	2.474E-06	6.945E-08	2.342E-08
Ni-63	O	9.048E-04	3.051E-04	8.127E-06	2.741E-06
Np-237	O	4.515E-06	1.523E-06	2.681E-08	9.041E-09
Pa-231*	O	2.866E-10	9.665E-11	5.246E-12	1.769E-12
Pb-210	O	3.221E-12	1.086E-12	1.919E-13	6.471E-14
Pd-107*	O	1.851E-06	6.242E-07	1.775E-08	5.986E-09
Pu-238	S	4.829E-02	1.628E-02	3.990E-03	1.345E-03
Pu-239	S	6.201E-03	2.091E-03	4.831E-04	1.629E-04
Pu-240	S	1.011E-02	3.409E-03	1.047E-03	3.530E-04
Pu-241	O	8.132E-01	2.742E-01	8.630E-03	2.910E-03
Pu-242	O	3.025E-05	1.020E-05	4.066E-07	1.371E-07
Pu-244	O	1.292E-11	4.357E-12	9.261E-14	3.123E-14

Table 1: The JN Standard Isotopic Mix (Continued)

Radionuclide	Basis	Ratioed Activity	Normalized Activity (1 Ci)	Ratioed Standard Deviation	Normalized Standard Deviation
Ra-226	O	1.494E-11	5.038E-12	5.753E-13	1.940E-13
Ra-228	O	1.995E-15	6.727E-16	5.133E-17	1.731E-17
Re-187*	O	2.909E-13	9.810E-14	2.500E-15	8.430E-16
Sb-125	O	4.581E-03	1.545E-03	1.376E-04	4.640E-05
Se-79	O	5.938E-06	2.002E-06	2.277E-08	7.678E-09
Si-32	O	6.174E-13	2.082E-13	1.013E-14	3.416E-15
Sm-147*	O	6.541E-11	2.206E-11	3.269E-13	1.102E-13
Sm-151	O	4.739E-03	1.598E-03	7.540E-06	2.543E-06
Sn-121m	O	9.678E-06	3.264E-06	7.716E-08	2.602E-08
Sn-126	O	1.206E-05	4.067E-06	7.395E-08	2.494E-08
Sr-90	O	6.563E-01	2.213E-01	3.127E-03	1.054E-03
Tc-99	O	1.884E-04	6.353E-05	6.950E-07	2.344E-07
Th-228*	O	4.600E-07	1.551E-07	2.466E-09	8.316E-10
Th-229	O	2.991E-12	1.009E-12	4.247E-14	1.432E-14
Th-230	O	3.070E-09	1.035E-09	6.482E-11	2.186E-11
Th-232	O	3.361E-15	1.133E-15	6.332E-17	2.135E-17
Tl-208*	O	1.654E-07	5.577E-08	8.890E-10	2.998E-10
U-232	O	4.620E-07	1.558E-07	2.626E-09	8.855E-10
U-233	O	5.377E-10	1.813E-10	7.163E-12	2.415E-12
U-234	O	1.727E-05	5.824E-06	3.050E-07	1.028E-07
U-235	O	2.523E-07	8.508E-08	7.807E-09	2.633E-09
U-236	O	3.344E-06	1.128E-06	3.620E-08	1.221E-08
U-238	O	4.899E-06	1.652E-06	2.829E-08	9.540E-09
Zr-93	O	2.774E-05	9.354E-06	1.153E-07	3.888E-08
Total		2.965E+00	1.000E+00		

- *The column labeled "Basis"contains either an "O" for ORIGEN2.1-derived data or an "S" for sample-derived data.*
- *Table 1 combines sample data and ORIGEN2.1 data. Because a "log-normal" distribution fit the sample data much better than a "normal distribution," the sample "log-normal" means and "log-normal" standard deviations were used. See the reference document containing the statistical analysis. Arithmetic means were used for the ORIGEN2.1 data because they were calculated values, unlike the sample data, which are experimental values. See the statistical analysis reference document.*
- *The isotopes marked with an "*" are not on the NRC license forEnvirocare. Because they are a very small percentage of the activity and because they were not confirmed by sample data, a check will be performed to assure that they do not represent in aggregate more than 5% of the activity. If this is true, they will be deleted from the records sent to Envirocare.*

2.0 JN STANDARD ISOTOPIC MIX

2.1 SITE HISTORY

The West Jefferson North site has four facilities (JN-1, JN-2, JN-3, and JN-4). Only three are covered by the BCLDP: JN-1, JN-2, and JN-3. JN-4 was cleaned and returned to Battelle for use on other projects. The project document TCP-98-03, "Building JN-1 Hot Cell Laboratory Acceptable Knowledge Document," summarizes all of the research conducted at the West Jefferson site and disposition of residual material from that research. The balance of this document deals with the majority of the JN-1 waste that is attributable to nuclear power reactor research. Several other BCLDP waste streams are handled separately.

The majority of waste at the West Jefferson North site is residual material from several years of research on irradiated fuel/Special Nuclear Material (SNM). In addition to the SNM nuclides, and transuranic (TRU) nuclides, decay products and activation products associated with the fuel are also found in varying amounts. While research was also conducted on other sources of radioactive material, those items are either segregated from the JN-1 waste or have been removed from the site completely.

A "JN Standard Isotopic Mix" is defined as a combination of samples and calculations further described as follows: 69 samples were taken throughout the accessible work areas of Building JN-1. The measured distribution of these 69 samples was compared with spent fuel modeled using the Oak Ridge Isotope Generation and Depletion (ORIGEN2.1) computer code. This code estimates the production and decay of fission and activation products of commercial nuclear power plant fuel. The ORIGEN2.1 input parameters of enrichment, burnup, and decay were varied to simulate fuel remnants at JN-1. (The ORIGEN2.1 Verification and Validation [V&V] Report identified in the "References" section documents the initial values for these three parameters and further describes the reasoning for assigning these values.) A series of software scripts and spreadsheets was used to reduce the sample data and the ORIGEN2.1 data so that they could be compared. A description of the various software applications used to define the JN Standard Isotopic Mix (called the software map) is included as Figure 1. The combined distribution of isotopes identified and estimated by sampling and modeling is called the JN Standard Isotopic Mix.

A given quantity of the JN Standard Isotopic Mix is then used as the radioactive material source with the Microshield computer shielding code to simulate external gamma ray interaction-rates for various package and form weights. These interaction rates are used to generate interaction rates-to-weight conversion equations for each package and waste form. The interaction rates are further used to calculate total activity content for individual packages and waste forms. From this, the level of TRU in nCi/g in each container is calculated. Those containers with levels greater than 100 nCi/g are considered TRU.

The direct sample, ORIGEN2.1 derived, and combined distribution means and uncertainties for each of the isotopes of interest are presented in Table 1. Refer to the "References" section for documents including the raw data for the 69 samples and the ORIGEN2.1 output tables. The radionuclides included in the mix were based on the following:

a. All mobile radionuclides in Hanford Site Solid Waste Acceptance Criteria (SWAC), except for H-3. (Past practices at Battelle make it likely that the H-3 has already been released from the waste.)

b. All radionuclides in the Hanford SWAC with a Category 1 waste limit, except Ti-44, Nb-91, Bi-207, and Bk-247, which were not present in samples or ORIGEN output.

c. All TRU radionuclides in Hanford SWAC with a half life greater than 20 years.

d. Sb-125, Cs-134, TI-208, and Th-228 added because they were present in samples.

e. Sum of Pu-239/240 from samples; Pu-239/240 allotted based on ORIGEN2.1 output.

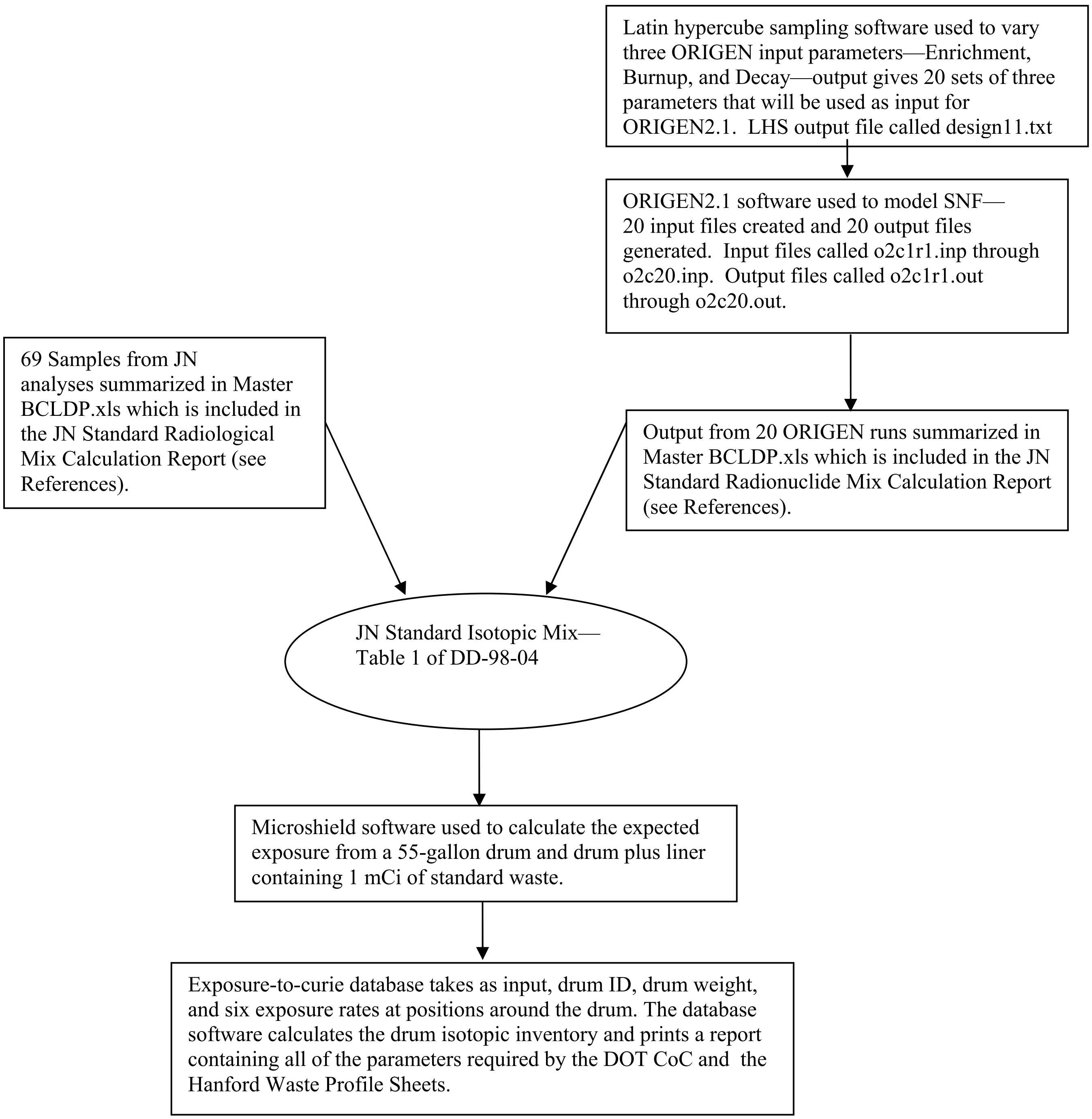
2.2 SOFTWARE MAP

Figure 1 demonstrates the relationship between all software packages used in developing DD-98-04. The software packages comprising the model are described in the order of use in the following sections. Please refer to the supporting documentation for specific assumptions and additional detail regarding software operation. Note, each piece of software output is verified and validated. The V&V report for each software output is filed with BCLDP project records.

2.3 SAMPLE-DERIVED ISOTOPIC MIX

Over a period of time, for various reasons, 69 samples have been collected throughout the JN-1 work area. A schematic showing JN-1 with the locations of 69 samples marked is included in the BCLDP project records (see JN Standard Mix Reference Document). This collection of 69 samples was used as the starting point for defining the JN Standard Isotopic Mix. If necessary, the mix will be re-evaluated according to BCLDP procedures (TC-AP-3.3, "Confirmation of TRU Waste Isotopic Mix") and re-established. The analysis of samples from JN-1 (Table 1) reveals the presence of Co-60, Cs-134, Cs-137, Eu-154, Am-241, Cm-244, Pu-238, Pu-239/240, U-234, U-238, Np-237, Sb-125, and Sr-90 in measurable quantities in at least one sample, with Cs-137 being dominant. Since Cs-137 is also the dominant isotope in the ORIGEN2.1 output, it is used as the basis for the standard mix specification. Ratios for all isotopes included in the standard mix but not identified in the samples are based on the ORIGEN2.1 calculations (daughter isotopes are routinely assumed to be in equilibrium with the parent and are not listed separately) and are then normalized to 1 Ci total waste.

Figure 1: Software Used to Develop DD-98-04



2.4 ORIGEN2.1-DERIVED ISOTOPIC MIX

Table 2 contains the ORIGEN2.1 input parameters and associated value ranges (low, center, and high) considered to represent most of the waste at the West Jefferson North site. Because the number of runs needed to represent the variations of the three parameters is so great, an alternate approach was used, reducing the number of runs to 20. This alternate approach is called Latin hypercube sampling, and it is further described in the subject document listed under the "References" section. Table 2 is further described in the ORIGEN2.1 V&V report included in the BCLDP project records and listed under the "References" section.

Table 2: ORIGEN2.1 Input Parameters

ORIGEN2.1 Parameter	Range of Values		
	Low	Center	High
Enrichment (atom % U-235)	2.0	2.8	4.0
Burnup (MCS/MTU)	15K	33K	45K
Decay (years post-irradiation)	13	17	30

In summary, four replicates of a five-sample Latin hypercube design were developed—using an S-Plus© version 2000 program written for this purpose and based on the procedure outlined in "Large Sample Properties of Simulations Using Latin Hypercube Sampling" by Michael Stein—thereby providing 20 analysis runs. The distribution of each parameter (Table 2) was divided into five partitions of equal probability. Latin hypercube sampling, then, ensures a random value of each partition is included in each of the five replicated designs, while minimizing the total number of required analysis runs. The resulting 20 (5 samples x 4 replicates) analysis runs are presented in Table 3.

Table 3: Experimental Design Parameter Values Used in ORIGEN2.1 Analysis Runs

Run No.	Enrichment	Burnup	Decay	Replicate	Sample
1	2.29	33.83	21.31	2	5
2	3.49	32.32	35.08	3	3
3	2.41	28.70	15.25	2	2
4	2.47	42.53	9.70	1	5
5	3.84	30.66	12.98	1	1
6	4.51	26.62	13.37	4	5
7	2.09	26.30	23.71	1	2
8	2.83	36.94	23.25	4	1
9	2.88	37.98	18.50	3	2
10	2.63	19.91	14.19	3	5
11	3.42	23.33	16.08	4	3
12	3.09	57.25	17.87	4	4
13	2.24	24.62	19.86	1	4
14	3.01	40.46	11.28	3	1

15	2.67	50.19	19.13	3	4
16	3.24	29.73	14.42	4	2
17	2.52	44.14	17.04	2	3
18	3.22	53.81	16.69	1	3
19	2.74	35.43	15.55	2	4
20	1.81	18.15	20.84	2	1

The mean result across the 20 ORIGEN2.1 runs (or equivalently, the mean of the mean results determined for the four replicated designs) calculated for each studied isotope represents its assumed normalized activity ratio (to Cs-137) in the standard isotopic distribution. Because of the extensive modeling capabilities of the ORIGEN2.1 code, the output contains many isotopes that are present in quantities below analytical detectability.

3.0 RH-TRU WASTE ACTIVITY CALCULATIONS

This section sets forth the method for identifying and quantifying the isotopic inventory for TRU waste containers. The waste addressed by this section is packaged into 55-gallon drums with steel liners. The liners are filled remotely inside the JN-1 hot cells. Once filled, each liner will receive two (2) designations. The first designation will be either CH or RH waste. Liners with contact exposure rates equal to or less than 200 mR/hr can be segregated as CH waste. The second designation will be either TRU waste or LLW. The waste is TRU if the concentration of transuranic isotopes is equal to or greater than 100 nCi/g. The U.S. Department of Energy identifies TRU waste as waste containing more than 100 nCi of alpha-emitting TRU isotopes per gram of waste, with half-lives greater than 20 years. CH-TRU will be set aside pending establishment of a method for assaying the waste. The current BCLDP strategy requires Waste Isolation Pilot Plant (WIPP) approved mobile vendor analysis for CH-TRU determination prior to shipment to the WIPP. The RH-TRU likely will be sent to Hanford for interim storage prior to disposal at the WIPP. Consequently, the Hanford Site SWAC is also followed.

3.1 RH-TRU WASTE—QUANTIFICATION OF RADIOACTIVE INVENTORIES

Liners packaging TRU waste with contact readings greater than 200 mR/hr are considered RH TRU waste and are further processed according to the following sections. For each RH-TRU waste-filled liner, two gamma exposure rates will be measured on opposite sides of the container at 1 meter from the mid-height of the side. Typically, this is done as the liner is pulled out of the hot cell. For these measurements, it is assumed that the background radiation level is insignificant in relation to the exposure rate produced by the waste container. The liner is then placed into a 55-gallon drum, weighed, and staged for shipment. The concentration of TRU is then calculated for each container according to the method described in Section 1.1. The following section describes the Microshield model for a 55-gallon drum liner of TRU waste.

3.2 MICROSHIELD MODEL FOR RH-TRU WASTE

The following input parameters describe the Microshield model used to calculate the exposure rate at 1 meter from 1 mCi of JN standard waste placed into a steel liner for a 55-gallon drum. The values are taken from the Microshield V&V report identified in the “References” section.

The steel liner has an outside diameter of 19.5 inches (49.53 cm) and a height of 32.25 inches (91.92 cm). The top and bottom of the liner’s circular steel plates have a thickness of 0.5 inches (1.27 cm). The liner wall is 0.105 inches (0.27 cm) thick. The liner has a tare weight of 160 pounds (72.7 kg). The liner can hold a volume of 39.5 gallons (149.7 liters) of waste.

For these calculations, the liner is full of waste and the radioactive material is uniformly dispersed throughout the waste. The waste matrix is assumed to be iron, and 10 waste densities range from 0.1515 to 2.4242 grams per cubic centimeter. These densities yield waste masses ranging from 50 to 800 pounds. The selection of iron as the waste matrix is not of great significance because the attenuation of the gamma rays of predominant interest is more influenced by the density of the material than by the specific composition or atomic number of the media.

It is also possible to take the exposure rate measurement after the liner has been placed in the 55-gallon container. The 55-gallon drum has a tare weight of 50 pounds (22.7 kg). The 55-gallon container was modeled as having an inside diameter of 22.25 inches (56.52 cm) and an interior height of 33.25 inches (84.46 cm). The 18-gauge steel used to fabricate the 55-gallon drum has a thickness of 0.0478 inches (0.12 cm).

The Microshield code uses the point kernel technique to calculate the exposure rate at a selected location from a distributed, multi-group source. The attenuation of the gamma rays and the build-up factors are computed for the source (the waste) and intervening shielding provided by packaging (the liner and, if appropriate, the 55-gallon drum). The results from this analysis are shown in Table 4 and in Figures 2 and 3. Note: 1 mCi was modeled in Microshield, and the results were then multiplied by 100 as shown in Table 4 and in Figures 2 and 3. This was done to ease data represention. See the MicroShield V&V report in the "References" section for the complete results.

Table 4: Exposure Rates (mR/hr) from 100 mCi Total Activity at 1 Meter From the Surface of the Waste Container at the Container Mid-Height

Waste Mass (lb)	Liner Plus Drum (mR/hr)	Liner Only (mR/hr)
50	12.41	13.50
100	11.52	12.58
150	10.63	11.63
200	9.78	10.72
250	8.99	9.87
300	8.28	9.1
350	7.64	8.41
400	7.07	7.78
450	6.56	7.23
500	6.11	6.73
550	5.71	6.29
600	5.35	5.90
650	5.02	5.54
700	4.73	5.22
750	4.47	4.94
800	4.24	4.68

Solving Equation 1 from Section 1.1 for A_{DRUM}, which is the term for the total activity in a given drum, gives the following:

$$A_{DRUM} = \dot{X}_{DRUM} \frac{A_{MS}}{\dot{X}_{MS}} \qquad \text{(Equation 2)}$$

Where

A_{DRUM}	=	Total activity in a drum
$\dot{X}_{DRUM}$	=	Actual measured exposure rate from a given drum
A_{MS}	=	Activity modeled by Microshield (1 mCi modeled) x 100 (to convert mCi to 100 mCi listed in Table 4)
$\dot{X}_{MS}$	=	Exposure rate calculated by MicroShield from a container of A_{MS} in a matrix of a given weight (From Table 4)

Once the total drum activity is calculated, it is further broken down into individual isotopic activities by multiplying it by the normalized values in Table 1.

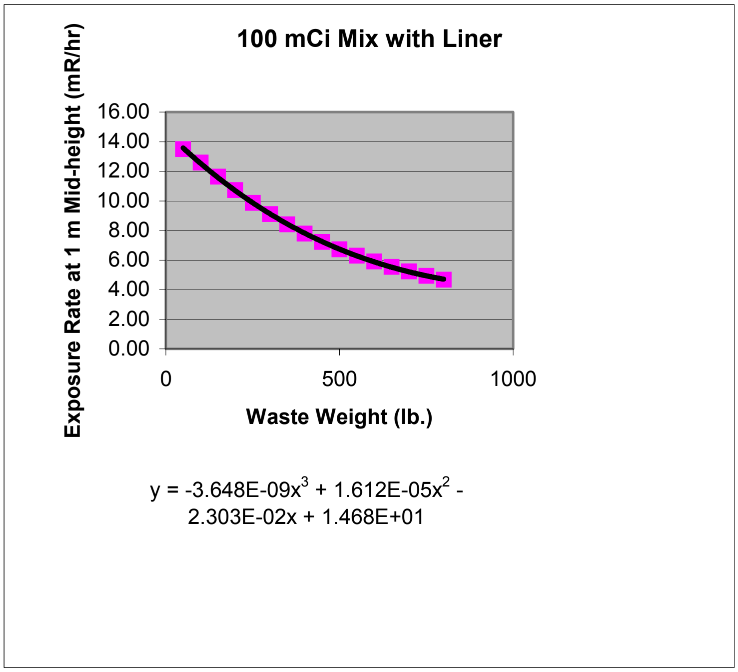


Figure 2: A Graphical Representation of the Data in Table 4—Exposure Rates (mR/hr) from 100 mCi Total Activity at 1 Meter From the Surface of the Waste Container at the Container Mid-Height - LINER ONLY

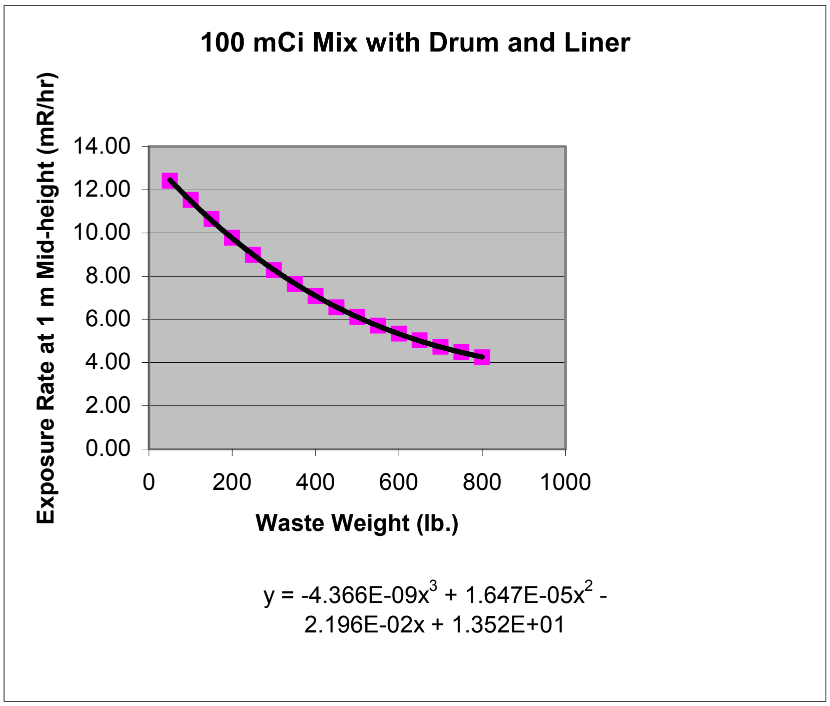


Figure 3: A Graphical Representation of the Data in Table 4—Exposure Rates (mR/hr) from 100 mCi Total Activity at 1 Meter From the Surface of the Waste Container at the Container Mid-Height - DRUM PLUS LINER

3.3 TREATMENT OF UNCERTAINTY FOR RH-TRU

Estimation of the total variance associated with the characterization of TRU waste has five components:

1. Uncertainty in the weight of the container being characterized;

2. Uncertainty in the assumed JN Standard Isotopic Distribution;

3. Uncertainty in measuring exposure rate;

4. Uncertainty in the dosage predicted (as a function of container configuration and weight) by the MicroShield software calculations; and,

5. Bias.

A detailed analysis of these sources of uncertainty is provided in the statistical analysis reference document. In summary, the relative uncertainty in the inventory of the individual isotopes is about 60 percent across the drum weights, with the exception of Co-60, which has relative uncertainty of nearly 110 percent. The consistency in the variance of isotopes other than Co-60 can be explained by noting that the variance of Co-60 contributes 80 percent or more to the total variance in an isotope's inventory. Over the range of weights of interest, the relative uncertainty in the TRU inventory is about 60 percent and the relative uncertainty in the total inventory is around 55 percent.

3.4 SOFTWARE QUALITY ASSURANCE

The use of software packages (i.e., ORIGEN2.1, Microshield, Excel spreadsheets) in the methodology has been determined under TCP-98-07, "Software Quality Assurance (QA) Plan for TRU Waste Certification Program (WCP)," to be compliant with the quality assurance requirements of the Quality Assurance Program Document (CAO-94-1012). All BCLDP TRU WCP software packages are used and maintained in accordance with TCP-98-07.

4.0 LLW WASTE ACTIVITY CALCULATIONS

This section addresses all waste other than that addressed by the TRU section. It sets forth the method for identifying and quantifying the isotopic inventory for LLW containers based upon the external gamma exposure rate. The waste addressed by this section is packaged into several different containers including

1. 55-gallon drums
2. B-25 boxes
3. Field sort waste bags
4. Other metal forms
5. B-12 boxes
6. IP-1 intermodal containers
7. IP-2 intermodal containers

4.1 QAD-CGGP-95.1 MODEL INPUT PARAMETERS

As was done for TRU waste described in the previous section, a known amount of JN waste was modeled for each of the container geometries. For the TRU waste, these calculations were performed using MicroShield; for LLW, these calculations were performed using the QAD-CGGP-95.1 computer program or MicroShield. As discussed in the MicroShield Version 5.05 Primitive Baseline, QAD and MicroShield yield equivalent results. For LLW, only the gamma-emitters Co-60 and Cs-137 were modeled. As discussed in the MicroShield RH-TRU V&V report, there is only a 3 percent difference between the results obtained using Co-60 and Cs-137 and those obtained using all the radionuclides in the JN Standard Isotopic Mix.

QAD-CGGP-95.1 or MicroShield calculations are performed for a range of representative weights for each package. The outputs for each package and form are subjected to regression analysis to obtain an equation describing the exposure rate as a function of weight. Once the external gamma exposure and the weight of the container are measured, a total container activity can be calculated. The total container activity is multiplied by the JN Standard Isotopic Mix fractions (Table 1) to get the inventory by radionuclide for the container.

4.2 SPECIFIC PACKAGE MODELS FOR LLW

4.2.1 55-Gallon Drums

The 55-gallon drum waste container was modeled as a cylinder with an inner radius of 11.25 inches (28.58 cm) and an inner height of 33.25 inches (84.46 cm). The wall of the drum was modeled as 0.0478 inch (0.12 cm) of iron. Exposure rates were determined at a point halfway up the drum, at a height of 16.63 inches (42.23 cm), and at a distance of 2 inches (5.08 cm) from the side of the drum.

Exposure rates were determined for waste weights that ranged from 50 pounds to 500 pounds (Table 5) for a drum containing 1 mCi of the JN Standard Isotopic Mix. The density of the waste material contained in the drum was derived from the calculated drum volume and waste weight.

Because the material placed in 55-gallon drums may vary from lab trash to construction debris, the material chosen for the waste matrix was iron.

The regression equation calculated from these data was

$$y = 1.769 - 1.618E\text{-}3\ x - 1.929E\text{-}6\ x^2 + 5.900E\text{-}9\ x^3 - 4.134E\text{-}12\ x^4 \qquad \text{(Equation 3)}$$

Where

x = Waste weight (lb)

y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

Table 5: Waste Weights and Exposure Rates in 55-Gallon Drums

Waste Weight (lb)	Exposure Rate (mR/hr)
50	1.6839
100	1.5937
150	1.5007
200	1.4086
250	1.3199
300	1.2359
350	1.1575
400	1.0849
450	1.0181
500	0.95692

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a 55-gallon drum.
Exposure rates are estimated at 2 inches from the side of drum, halfway up the drum.

4.2.2 B-25 Boxes

B-25 boxes are rectangular sheet-metal bins generally used for packaging irregularly shaped or bulky materials for shipment and/or disposal. The inner dimensions of a standard B-25 box are 6 feet (182.88 cm) wide, by 3 feet 11 inches (119.38 cm) tall, by 3 feet 9 inches (114.3 cm) deep, for a total actual volume of 88 cubic feet.

Box walls were modeled as being 0.105 inches (0.266 cm) thick. Exposure rate calculations were performed for contact radiation level measurements taken halfway up at the midpoint of the long side of the box. The model also included a 2-inch (5.08-cm) air gap between the side of the box and the measurement point since the distance from the outside of a box to the centerline of the ion chamber used is 2 inches.

For the JN site, a typical box weighs approximately 4,500 lbs. Cases were run ranging from 1,000 pounds to 9,000 pounds. The weight limit for B-25 containers being sent to Envirocare is 9,000 pounds. The weight limit for containers being sent to Hanford is 7,000 pounds. Because objects placed in B-25 boxes are often metal, the material chosen for the waste matrix was iron.

Several cases were also run to evaluate the effect of changes in waste material composition on exterior B-25 box exposure rates. Previous modeling demonstrated that changing the waste material to concrete from iron resulted in a change in external package exposure rates of approximately 8 percent. This effect is sufficiently small that the model may be considered valid for B-25 boxes containing metal scrap, concrete debris, or a combination of these materials.

B-25 boxes present a special case in that they must be foam-filled to eliminate void space. To preclude the possibility that a full box will exceed applicable activity limits, and to ensure that a box will not have to be repackaged to meet these limits, QAD-CGGP-95.1 calculations were made for full and ¾, ½, and ¼ full B-25 boxes (see Tables 6 through 9). Cases run with varying densities (i.e., from 1,000 pounds to 9,000 pounds) indicated some variation in expected contact exposure rate with weight.

The regression equation for the full box was

$$y = 4.561\text{E-}1 - 1.201\text{E-}4\ x + 1.695\text{E-}8\ x^2 - 1.235\text{E-}12\ x^3 + 3.639\text{E-}17\ x^4 \qquad \text{(Equation 4)}$$

The regression equation for the 3/4 full box was

$$y = 3.772\text{E-}1 - 1.227\text{E-}4\ x + 2.210\text{E-}8\ x^2 - 2.080\text{E-}12\ x^3 + 7.944\text{E-}17\ x^4 \qquad \text{(Equation 5)}$$

The regression equation for the 1/2 full box was

$$y = 2.280\text{E-}1 - 1.201\text{E-}4\ x + 3.388\text{E-}8\ x^2 - 4.937\text{E-}12\ x^3 + 2.908\text{E-}16\ x^4 \qquad \text{(Equation 6)}$$

The regression equation for the 1/4 full box was

$$y = 1.492\text{E-}1 - 1.279\text{E-}4\ x + 6.341\text{E-}8\ x^2 - 1.669\text{E-}11\ x^3 + 1.788\text{E-}15\ x^4 \qquad \text{(Equation 7)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

Table 6: Waste Weights and Exposure Rates for Full B-25 Boxes

Waste Weight (lb)	Exposure Rate (mR/hr)
1,000	3.51E-01
1,500	3.10E-01
2,000	2.74E-01
2,500	2.44E-01
3,000	2.18E-01
3,500	1.96E-01
4,000	1.77E-01
4,500	1.61E-01
5,000	1.48E-01
5,500	1.36E-01
6,000	1.26E-01
6,500	1.17E-01
7,000	1.09E-01
7,500	1.02E-01
8,000	9.63E-02
8,500	9.08E-02
9,000	8.59E-02

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a B-25 box. Exposure rates are estimated at 2 inches from the side of the box.

Table 7: Waste Weights and Exposure Rates for 3/4 Full B-25 Boxes

Waste Weight (lb)	Exposure Rate (mR/hr)
750	2.97E-01
1,125	2.65E-01
1,500	2.36E-01
1,875	2.12E-01
2,250	1.91E-01
2,625	1.73E-01
3,000	1.58E-01
3,375	1.45E-01
3,750	1.34E-01
4,125	1.24E-01
4,500	1.16E-01
4,875	1.08E-01
5,250	1.01E-01
5,625	9.55E-02
6,000	9.02E-02
6,375	8.54E-02
6,750	8.11E-02

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a B-25 box. Exposure rates are estimated at 2 inches from the side of the box.

Table 8: Waste Weights and Exposure Rates for 1/2 Full B-25 Boxes

Waste Weight (lb)	Exposure Rate (mR/hr)
500	1.76E-01
750	1.55E-01
1,000	1.37E-01
1,250	1.22E-01
1,500	1.09E-01
1,750	9.78E-02
2,000	8.85E-02
2,250	8.06E-02
2,500	7.38E-02
2,750	6.80E-02
3,000	6.29E-02
3,250	5.85E-02
3,500	5.46E-02
3,750	5.12E-02
4,000	4.81E-02
4,250	4.54E-02
4,500	4.30E-02

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a B-25 box. Exposure rates are estimated at 2 inches from the side of the box.

Table 9: Waste Weights and Exposure Rates for 1/4 Full B-25 Boxes

Waste Weight (lb)	Exposure Rate (mR/hr)
250	1.21E-01
375	1.09E-01
500	9.92E-02
625	9.02E-02
750	8.24E-02
875	7.56E-02
1,000	6.97E-02
1,125	6.46E-02
1,250	6.01E-02
1,375	5.62E-02
1,500	5.27E-02
1,625	4.96E-02
1,750	4.68E-02
1,875	4.43E-02
2,000	4.20E-02
2,125	4.00E-02
2,250	3.81E-02

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a B-25 box. Exposure rates are estimated at 2 inches from the side of the box.

4.2.3 Field Sort Waste Bags

Two sizes of full, individual, field sort waste bags were considered: a small bag 12 inches in diameter and 15 inches high, and a tall bag 18 inches in diameter and 27 inches high. Cases were calculated for full small bags at incremental weights of 1, 4, 8, 12, 16, 20, 24, 26, 28, and 30 pounds; and for full tall bags at incremental weights of 5, 10, 15, 20, 25, 30, 35, 40, 45, 50, 55, 60, 65, and 70 pounds. In all cases the source material was modeled as cellulose because the bags generally contain light materials (e.g., paper products, latex gloves, and various cloth items). Additionally, the measurement point was set at 1 inch from the surface at the centerline (i.e., mid-plane) of the bag to represent the distance from the outside of a bag to the center of the detector.

Tables 10 and 11 contain the results of QAD-CGGP-95.1 calculations for individual trash bag cases. Curve-fitting techniques were applied to these results used to determine equations to describe external exposure rates as a function of bag weight.

The regression equation calculated for the small bags was

$$y = 7.227 - 1.757E\text{-}2\ x + 4.243E\text{-}4\ x^2 - 2.331E\text{-}5\ x^3 + 3.108E\text{-}7\ x^4 \qquad \text{(Equation 8)}$$

The regression equation calculated for the tall bags was

$$y = 3.208 - 2.968E\text{-}3\ x + 2.601E\text{-}5\ x^2 - 6.034E\text{-}7\ x^3 + 3.319E\text{-}9\ x^4 \qquad \text{(Equation 9)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

Table 10: Waste Weights and Exposure Rates for Small Field Sort Bags

Waste Weight (lb)	Exposure Rate (mR/hr)
1	7.21E+00
4	7.16E+00
8	7.10E+00
12	7.04E+00
16	6.98E+00
20	6.91E+00
24	6.83E+00
26	6.79E+00
28	6.75E+00
30	6.70E+00

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a small field sort bag.

Table 11: Waste Weights and Exposure Rates for Tall Field Sort Bags

Waste Weight (lb)	Exposure Rate (mR/hr)
5	3.19E+00
10	3.18E+00
15	3.17E+00
20	3.15E+00
25	3.14E+00
30	3.13E+00
35	3.11E+00
40	3.10E+00
45	3.09E+00
50	3.07E+00
55	3.05E+00
60	3.04E+00
65	3.02E+00
70	3.00E+00

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a tall field sort bag.

4.2.4 Other Metal Forms

The activity content of a number of waste components cannot be adequately calculated by the models described previously. Among these waste components are ventilation ducts, pumps, valves, and motors. To better calculate the activity content of items such as these, two general models and spreadsheets were developed. The models developed are small and large metal (iron) cylinders, which are discussed below in detail. Both models are fixed in size, but cover a range of weights, and therefore densities, to provide a means to calculate activities contained in a broad range of real objects.

4.2.4.1 Small Metal Cylinder

The small metal model is a right circular cylinder 6 inches in radius and 15 inches high. The exposure measuring point is separated from the source by a 1-inch air gap, which simulates the centerline of the indicated ion chamber detector.

QAD-CGGP-95.1 calculations were performed for this model with net weights of 5, 10, 15, 20, 25, 30, 40, 50, and 60 pounds. The radiation source for these calculations was 1 mCi of the JN Standard Isotopic Mix. The point at which the exposure was modeled is the vertical centerline of the cylinder.

Table 12 contains the results of QAD-CGGP-95.1 calculations. Curve-fitting techniques were applied to these results to determine equations to describe external exposure rates as a function of weight.

Table 12: Waste Weights and Exposure Rates for Small Metal Cylinders

Waste Weight (lb)	Exposure Rate (mR/hr)
5	7.10E+00
10	6.98E+00
15	6.84E+00
20	6.71E+00
25	6.57E+00
30	6.42E+00
40	6.14E+00
50	5.85E+00
60	5.57E+00

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a small metal cylinder.

The regression equation calculated for the small metal waste cylinders was

$$y = 7.225 - 2.352E\text{-}2\ x - 1.464E\text{-}4\ x^2 + 1.448E\text{-}6\ x^3 - 2.486E\text{-}9\ x^4 \qquad \text{(Equation 10)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

4.2.4.2 Large Metal Cylinder

The large metal model is a right circular cylinder 12 inches in diameter by 36 inches high. The exposure measuring point is separated from the source by a 1-inch air gap, which simulates the measuring point of the radiation survey instrument.

QAD-CGGP-95.1 calculations were performed for this model with net weights from 10 pounds to 100 pounds in 10-pound increments. The radiation source for these calculations was 1 mCi of the JN Standard Isotopic Mix. The point at which the exposure was modeled is the vertical centerline of the cylinder.

Table 13 contains the results of QAD-CGGP-95.1 calculations. Curve-fitting techniques were applied to these results to determine equations to describe external exposure rates as a function of weight.

Table 13: Waste Weights and Exposure Rates for Large Metal Cylinders

Waste Weight (lb)	Exposure Rate (mR/hr)
10	3.97E+00
20	3.89E+00
30	3.81E+00
40	3.73E+00
50	3.64E+00
60	3.55E+00
70	3.46E+00
80	3.37E+00
90	3.28E+00
100	3.19E+00

Note: Exposure rates are for 1 mCi of JN Standard Isotopic Mix in a large metal cylinder.

The regression equation calculated for the large metal cylinders was

$$y = 4.050 - 7.329E\text{-}3\ x - 2.439E\text{-}5\ x^2 + 1.275E\text{-}7\ x^3 - 7.139E\text{-}11\ x^4 \qquad \text{(Equation 11)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

4.2.5 B-12 Boxes

The B-12 box was modeled as a rectangular solid with dimensions 6 feet wide by 2 feet tall by 4 feet deep. The wall of the box was modeled as 12-gauge steel, 0.1046 inch thick. Exposure rates were determined at a point halfway up the box (12 inches) and halfway from the edge of the box (36 inches) and at distance of 2 inches from the side of the box.

Exposure rates were determined using MicroShield for waste weights that ranged from 500 pounds to 9,000 pounds (Table 14) for a box containing 1 mCi of the JN Standard Isotopic Mix. The density of the waste material contained in the box was derived from the calculated box volume and waste weight. The material chosen for the waste matrix was lead because lead is often placed in the boxes.

Table 14: Waste Weights and Exposure Rates for B-12 Boxes

Waste Weight (lb)	Exposure Rate (mR/hr)
500	3.672E-1
1,000	2.740E-1
1,500	2.163E-1
2,000	1.777E-1
2,500	1.503E-1
3,000	1.299E-1
3,500	1.143E-1
4,000	1.019E-1
4,500	9.188E-2
5,000	8.360E-2
5,500	7.665E-2
6,000	7.074E-2
6,500	6.565E-2
7,000	6.124E-2
7,500	5.737E-2
8,000	5.395E-2
8,500	5.091E-2
9,000	4.819E-2

Note: Exposure rates are for 1 mCi of the JN Standard Isotopic Mix in a B-12 box.

The regression equation calculated from these data was

$$y = 4.629\text{E-}1 - 2.320\text{E-}4\ x + 5.666\text{E-}8\ x^2 - 6.407\text{E-}12\ x^3 + 2.680\text{E-}16\ x^4 \qquad \text{(Equation 12)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

4.2.6 IP-1 Intermodal "Rolloff" Containers

The 25.4-cubic-yard IP-1 intermodal container was modeled as a rectangular solid with dimensions 230 inches wide by 61 inches tall by 85 inches deep. The half-full IP-1 container was assumed to be filled to a height of 30.5 inches. The wall of the IP-1 container was modeled as 10-gauge steel, 0.1345 inch thick. Exposure rates were determined at a point halfway up the waste (15.25 inches) and halfway from the edge of the box (115 inches) and at distance of 2 inches from the side of the container for a half-full IP-1 container. For a full IP-1 container, exposure rates were determined at a point halfway up the waste (30.5 inches) and halfway from the edge of the box (115 inches) and at a distance of 2 inches from the side of the container.

Exposure rates were determined using MicroShield for waste weights that ranged from 6,000 pounds to 37,000 pounds (Tables 15 and 16) for an IP-1 container containing 1 mCi of the JN

Standard Isotopic Mix. The density of the waste material was derived from the calculated IP-1 volume and waste weight. The material chosen for the waste matrix was concrete because building debris and soil are often placed in the containers.

Table 15: Waste Weights and Exposure Rates for Half Full IP-1 Intermodal Containers

Waste Weight (lb)	Exposure Rate (mR/hr)
6,000	7.829E-2
9,100	6.214E-2
12,200	5.144E-2
15,300	4.383E-2
18,400	3.812E-2
21,500	3.369E-2
24,600	3.015E-2
27,700	2.725E-2
30,800	2.484E-2
33,900	2.281E-2
37,000	2.107E-2

Note: Exposure rates are for 1 mCi of the JN Standard Isotopic Mix in a half-full IP-1 intermodal container.

Table 16: Waste Weights and Exposure Rates for Full IP-1 Intermodal Containers

Waste Weight (lb)	Exposure Rate (mR/hr)
6,000	7.723E-2
9,100	6.397E-2
12,200	5.383E-2
15,300	4.611E-2
18,400	4.013E-2
21,500	3.542E-2
24,600	3.163E-2
27,700	2.852E-2
30,800	2.594E-2
33,900	2.377E-2
37,000	2.191E-2

Note: Exposure rates are for 1 mCi of the JN Standard Isotopic Mix in a full IP-1 intermodal container.

For a half-full IP-1 container, the regression equation calculated from these data was

$$y = 1.270E\text{-}1 - 1.075E\text{-}5\ x + 4.986E\text{-}10\ x^2 - 1.156E\text{-}14\ x^3 + 1.039E\text{-}19\ x^4 \qquad \text{(Equation 13)}$$

For a full IP-1 container, the regression equation calculated from these data was

$$y = 1.145E\text{-}1 - 7.785E\text{-}6\ x + 2.948E\text{-}10\ x^2 - 5.825E\text{-}15\ x^3 + 4.641E\text{-}20\ x^4 \qquad \text{(Equation 14)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

4.2.7 IP-2 Intermodal "Sealand" Containers

The 37-cubic-yard IP-2 intermodal container (Sealand Container) was modeled as a rectangular solid with dimensions 227 inches wide by 91 inches tall by 88 inches deep. The half-full IP-2 container was assumed to be filled to a height of 45.5 inches. The wall of the IP-2 container was modeled as 10-gauge steel, 0.1345 inch thick. Exposure rates were determined at a point halfway up the waste (22.75 inches) and halfway from the edge of the container (113.5 inches) and at a distance of 2 inches from the side of the container for a half full IP-2 container. For a full IP-2 container, exposure rates were determined at a point halfway up the waste (45.5 inches) and halfway from the edge of the box (113.5 inches) and at a distance of 2 inches from the side of the container.

Exposure rates were determined using MicroShield for waste weights that ranged from 6,000 pounds to 34,000 pounds (Tables 17 and 18) for an IP-2 container containing 1 mCi of the JN Standard Isotopic Mix. The density of the waste material was derived from the calculated IP-2 volume and waste weight. The material chosen for the waste matrix was concrete because building debris and soil are often placed in the containers.

For a half full IP-2 container, the regression equation calculated from these data was

$$y = 1.228E\text{-}1 - 9.652E\text{-}6\ x + 4.289E\text{-}10\ x^2 - 9.861E\text{-}15\ x^3 + 9.021E\text{-}20\ x^4 \qquad \text{(Equation 15)}$$

For a full IP-2 container, the regression equation calculated from these data was

$$y = 9.508E\text{-}2 - 4.773E\text{-}6\ x + 1.210E\text{-}10\ x^2 - 1.389E\text{-}15\ x^3 + 4.361E\text{-}21\ x^4 \qquad \text{(Equation 16)}$$

Where

x = Waste weight (lb)
y = Exposure rate (mR/hr for 1 mCi of JN Standard Isotopic Mix)

**Table 17: Waste Weights and Exposure Rates for
Half-Full IP-2 Intermodal Containers**

Waste Weight (lb)	Exposure Rate (mR/hr)
6,000	7.833E-2
8,800	6.474E-2
11,600	5.474E-2
14,400	4.723E-2
17,200	4.143E-2
20,000	3.683E-2
22,800	3.310E-2
25,600	3.002E-2
28,400	2.744E-2
31,200	2.525E-2
34,000	2.337E-2

Note: Exposure rates are for 1 mCi of the JN Standard Isotopic Mix in a half-full IP-2 intermodal container.

**Table 18: Waste Weights and Exposure Rates for
Full IP-2 Intermodal Containers**

Waste Weight (lb)	Exposure Rate (mR/hr)
6,000	7.048E-2
8,800	6.159E-2
11,600	5.390E-2
14,400	4.743E-2
17,200	4.207E-2
20,000	3.761E-2
22,800	3.388E-2
25,600	3.075E-2
28,400	2.810E-2
31,200	2.583E-2
34,000	2.388E-2

Note: Exposure rates are for 1 mCi of the JN Standard Isotopic Mix in a full IP-2 intermodal container.

5.0 DIRECT SAMPLE ANALYSIS WASTE STREAMS

Fuel Pool Waste—Waste from the JN-1 fuel pool will be characterized based upon direct sample results. Pool wastes currently drummed and stored pending shipment were individually characterized due to solubility differences between various nuclides in the mix. Cesium, upon which this modeling methodology depends, is highly soluble; while the transuranic isotopes and others tend to be largely insoluble. This inconsistency rendered cesium-based modeling unsuitable for any waste stream in which solubility is a major concern. Therefore, all pool-related wastes will be characterized individually. This is outside the scope of the methodology in this document.

Plutonium Wastes from Buildings JN-2 and JN-4—Plutonium processing was conducted at one time in Building JN-4. After that research was completed, the waste was collected and stored in Buildings JN-2 and JN-3. It has been kept separate from the other JN waste streams and will be addressed by a future revision of this document.

6.0 REFERENCES

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   f. Microshield
      1. Software Documentation (called Primitive Baseline according to BCLDP Software QA Plan) transmitted to BCLDP files by Jim Eide on October 15, 2002
      2. V&V Report by A. Plummer
   g. Dose-to-curie Database
      1. Software Documentation transmitted to BCLDP project records by J. Eide November 2002
      2. V&V Report by S. Maheras
3. "Hanford Solid Waste Acceptance Criteria (SWAC)", HNF-EP-0063.

4. U.S. Department of Transportation Certificate of Compliance for the Chem-Nuclear Systems (CNS) 10-160B Container.

5. Stein, Michael. "Large Sample Properties of Simulations Using Latin Hypercube Sampling," *Technometrics,* May 1987, 29(2):143-151.

6. Battelle Memorial Institute, "Software Quality Assurance (QA) Plan for TRU WCP," TCP-98-07, Battelle Memorial Institute, BCLDP, Columbus, Ohio.

7. Carlsbad Area Office Quality Assurance Program Document, CAO-94-1012.